TO:

Allen R. Blough, September 30, 1986

Reactor Projects, Section 1A,

Division of Reactor Projects,

NRC Region 1,

631 Park Avenue, King of Prussia, Pa.

Dear Mr. Blough:

As I mentioned to you in the beginning of the Unit 1 Salp Meeting, I'm forewarding the identifying data on the Day 31 page from the B&W Trial records. The page in question is the R. L. Long Exibits...Exibit Number 692. According to the cc s in the accompanying cover letter, a copy of the exibit, a sequence of events, and the letter were sent to a B.H. Grier of Region 1's Office of Inspection and Enforcement, so you may have a full copy somewhere in Region 1..it also went to an S. A. Varga in D.C.. Enclosed is a xerox of the cover letter.

The Day 31 events did raise some additional questions regarding steam generators when primary to secondary contamination of note occurs....who authorizes the continued use...or putting back in use of said generators..and what are the factors governing such decisions. If I understood correctly...there has been some initial study or decision underway which advocates the steaming on the damaged generator, rather than its isolation...and that by so doing...releases would purportedly be lower to the environment outside the plant. Could you comment on that again, or could Mr. Conte..I believe he mentioned the study in progress. I may not have understood the commerts made re. the proceedures to be followed regarding significant steam tube leaks, or steam tube failures. Of course, it must be assumed that all tube failures would be significant.

Also note on day 32, the B Generator was apparently isolated again, because measurements on the station vent monitor increased. Actually....the Turbine Bypass valves were isolated. Would this mean B was isolated totally re. releases? The readings on the 32nd day were felt to have been due to the changing of a charcoal filter; however, one might still wonder about the cause of the increased levels of radioactive materials present in THI 2 on the 31st day of the accident. I Am also wondering what stack monitor read in the 31st day, also if some defect exists, or potential for added releases is present in condensore that would have caused releases like those noted on Day 31?

Sincerely,

Deborah Davenport

Deborah Davenport,
1802 Harlet Street,
Camp Hill, Penna. 17011
1- 717- 763-9552

**

Also note, are copies of my letters, and the xeroes are well being sent to the PDRs and the service lists, ; have not been able to get to document room, and was not certain if all data was being sent through. If it has not included, I would like to request that all letters and xeroxes be sent out to all of lists etc... if so I thank you retrospectively.

X

MAR 18 1980 R. Long

MetEd / GPU

Metropolitan Edison Company
Post Office Box 542
Reading Pennsylvania 19640
215 929-3601

Writer's Direct Dial Number

Mr. B. H. Grier, Director
Office of Inspection & Enforcement
Region 1
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

In our letter of July 16, 1979, we indicated that by July 20, 1979, we would submit a detailed Sequence of Events covering the period through April 30, 1979. Enclosed please find that Sequence of Events. Please add this attachment to Section I of our July 16, 1979 submittal.

Sincerely,

/s/ J. G. Herbein

J. G. Herbein
Vice President-Generation

JGH:RAL:tas

Enclosure

cc: Director of Nuclear Reactor Regulation
Attn: S. A. Varga
Light Water Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

B&W ~~Deft~~ Exh. For ID 692
~~Pl. Exh in EV~~ 5-19-82
Harvey B. Kramer CSR
Doyle Reporting, Inc.

GPU B&W RL CONTROL TMI

1313 8 0 1

Inter-Office Memorandum

GPU Service

Date: JULY 17, 1979

Subject: PRELIMINARY ANNOTATED SEQUENCE OF EVENTS, MARCH 29, 1979 THRU APRIL 30, 1979, REV. 0

To:

Location: TMI

MESSRS.
R. ARNOLD
M. BEZILLA
Z. BLAKE
T. BROUGHTON
J. DANIELS
H. DIECKAMP
DRMC
J. HERBEIN
R. HEWARD
R. KRAYER
W. LAVALLEE (EPRI)
J. LOGAN
R. LONG
G. MILLER
W. MOREAU (2)
M. MORRELL
L. ROGERS
B. RUSCHE
J. SEELINGER
J. THORPE
A. TSAGGARIS
T. VANWYBECK
E. WALLACE (2)
R. WILLIAMS
R. WILSON
K. WOODWARD
R. ZELBOSKI (EPRI)

Enclosed is the 7/16/79 draft of Rev. 0 to the "Preliminary Annotated Sequence of Events, March 29, 1979 thru April 30, 1979". Should you have any comments, please return them to me no later than August 13, 1979.

Robert L. Long
Accident Assessment Documentation

RLL:bas

Enclosure

~~Read this page until you understad i~~
~~2-Read this page until you understand it~~

SUMMARY OF MAJOR EVENTS OCCURRING ON DAY 31

1. Pressurizer Level Transmitter (RC-1-LT3) failed. The operator commenced Emergency Procedure EP-21, "Total Loss of Pressurizer Level Indication" to determine level.

2. In preparation for natural circulation, Steam Generator B was steamed to the condenser using the Turbine Bypass Valves at 25% open. Due to this steaming, increases in airborne radioactivity were detected. Respirators were required in the Control Room for several minutes and in the Turbine Building for between three to four hours. All unnecessary personnel were evacuated from Unit 2 Control and Turbine Buildings. Initially it was thought that the high airborne activity was Iodine. Later events showed that the activity was mostly Xenon 133 with traces of Xenon 131 and 135 which had saturated the SAM-2 detectors and gave a false Iodine indication.

The transition to natural circulation was accomplished at 1408 hours by stopping Reactor Coolant Pump 2A (RC-P-2A). Steam Generators A and B were used to pass steam to the condenser through the Turbine Bypass Valves (MS-V15A, 15A) which were 100 % and 25% open, respectively. Natural circulation and cooldown at 5° F per hour went smoothly with dT_A = 10° and dT_B = 14°. Due to low steaming rate on Steam Generator B the dT_B ultimately increased to approximately 44°. Because of the degradation of the Pressurizer and Steam Generator level instrumentation the transition to natural circulation was initiated ahead of schedule.

Note: dT_A and dT_B represent the temperature difference across the primary sides of Steam Generators "A" and "B", respectively.

06/22/79
Rev. 0

04/23/79
Day 22

SUMMARY OF MAJOR EVENTS OCCURRING ON DAY 22

1. At 0110 hours the Turbine Bypass Valves for Steam Generator 3 (MS-7-253 and 263) were isolated. This was initiated when the Station Vent Radiation Monitor HPR-219 increased. It was later determined that the increase was due to changing the charcoal filters. This was scheduled to be done at midnight (0000 hours), but due to delays it was done around 0100 hours.

2. Pressurizer level was being calculated using Emergency Procedure EP-21, "Total Loss of Pressurizer Level Indication." This value was being compared with Pressurizer Level Transmitter (RC-1-LT3) which was back in service. The decision was made not to use the back-up Heise gauge due to its poor correlation to the measured level.

3. Average Reactor Coolant temperature decreased from 182F to 175F during the day as natural circulation continued.

4. Approximately 2940 gallons of water were added to the Makeup Tank (MU-T-1).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

JAN 16 1987

Docket No. 50-289

Ms. Deborah Davenport
1802 Market Street
Camp Hill, Pennsylvania 17011

Dear Ms. Davenport:

This is in response to your letters, dated September 15 and 30, 1986, to me. We have summarized our responses to questions in your letters in the attachment. Your letters are attached for completeness and forwarding to the docket rooms. If you have questions, please contact either Rich Conte (215-337-5146) or me.

Sincerely,

Allen R. Blough, Chief
Projects Branch No. 1
Division of Reactor Projects

Attachments:
As Stated

cc:
PDR
LPDR
Senior Resident Inspector (TMI-1)

bcc:
A. Blough
R. Conte
W. Baunack
K. Abraham

 4 pp.

ATTACHMENT
RESPONSES TO D. DAVENPORT LETTERS

A. Letter dated September 15, 1986

1. In reference to releases during a steam tube failure, provide copies of any evaluation or procedures that are available.

Response

The design basis steam generator tube rupture event was evaluated in relation to the Kinetic Expansion repair process for the TMI-1 steam generator tubes. The results of that review are documented in NUREG 1019 (Section 4) and Supplement 1 (Section 4), which should be available for review in the Local Public Document Room (LPDR). Specifically Sections 4.3 of both documents address this topic along with licensee procedures for handling such events. Further, the licensee's methodology for handling such events are being generically reviewed under TMI Task Action Plan (TAP) Item No. I.C.1, "Accident and Procedure Review." Recent licensee submittals on this topic should also be available for review in the LPDR.

2. Several sections of the letter reiterate the request for a filter on the condenser off-gas system.

Response

This request was answered in a letter, dated November 26, 1986, from H. Denton, NRR, to you.

B. Letter dated September 30, 1986

1. Comment on the steaming of, rather than isolating, a damaged steam generator (upon tube rupture) and how that lowers releases to the environment.

Response

If a steam generator tube rupture should occur along with multiple tube ruptures, the best way to stop the ensuing primary to secondary leakrate is to cooldown and depressurize the reactor coolant system The quickest way to do that within cooldown limits is to use both steam generators rather than other slower methods such as steaming one undamaged steam generator or using the RCS feed and bleed methodology. Use of the steam generator is complicated by releases of radioactivity eventually through the condenser off-gas system or through the steam generator safety/relief valves if the leak is large enough to cause RCS pressure (if above 1040 psig) to be applied to the secondary side of the steam generator (relief valve settings at 1040-1100 psig). Releases are inevitable but they are within

design basis limits (10 CFR 100) for the design basis event (single tube failure). The licensee is also prepared for the beyond-design-basis event (multiple tube failure) in order to minimize releases to the public. For any of the events noted above and those releases, there will be a dose rate to the public. The licensee's approach is to shorten, as much as possible, the length of time to stop the primary to secondary leakage, so that the total dose to the public will be lessened. This approach is under generic review by NRC as stated in paragraph A.1 above.

Briefly, upon detection of primary to secondary leak rate of 1 to 50 gpm, licensee procedures require the plant to be shutdown and immediately cooled to below 540° F (from the normal 579° F); and an orderly cooldown and depressurization is commenced. The affected steam generator is isolated if the off-site dose projection approaches 50 mR/hr whole body or 250 mR/hr thyroid dose rate.

2. In reference to day 32 after the TMI-2 accident, Ms. Davenport asks that, if the turbine bypass was isolated, would that mean the "B" steam generator would be isolated totally with respect to releases? For day 31, what were the stack monitor releases at TMI-2?

Response

In response to your specific question about the turbine bypass valve, a number of valves in the main steam and feedwater system must be shut to isolate a steam generator; e.g., main steam isolation valves and feedwater isolation/block valves. Even with this isolation, there is a possible release path from an isolated steam generator through the steam generator safety/relief valve, which could actuate automatically on a high pressure situation in the steam generator (at approximately 1040-1100 psig). Even with the design, as noted in the NRR letter to you dated November 26, 1986, the TMI-1 design meets the objective of 10 CFR 50 Appendix I.

The TMI-2 accident and subsequent days after that event have been extensively reviewed and evaluated by NRC staff. It would not be an appropriate expenditure of NRC staff resources to revisit and answer specific questions about those events unless a new safety question needs to be addressed. Your request for a filtration system on the condenser off-gas system, has been reviewed by NRC staff and answered by Mr. Denton's letter.

3. Are copies of my letters being sent to PDR's and service lists? If not, I am requesting that they be sent to all lists, etc.

Response

All of your letters to Region I are being sent to the PDR and LPDR, either separately or by attachment to the NRC staff response letters. Any discrepancies in that regard should be brought to our attention. With respect to the service list, it is not the policy of Region I to serve this type of correspondence on the Board or parties.